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ROBERT C. MECREDY
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November 20, 2000

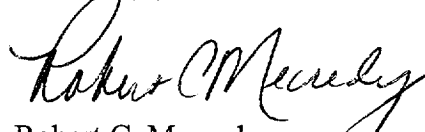
U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I
Washington, D.C. 20555

Subject: LER 2000-005, Loss of "B" Condenser Circulating Water Pump Results in
Manual Reactor Trip
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

The attached Licensee Event Report LER 2000-005 is hereby submitted in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (iv), which requires a report of, "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)".

Very truly yours,



Robert C. Mecredy

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
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LICENSEE EVENT REPORT (LER)

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digits/characters for each block)

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DOCKET NUMBER (2)

05000244

PAGE (3)

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TITLE (4)

Loss of "B" Condenser Circulating Water Pump Results in Manual Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	21	2000	2000	005	00	11	20	2000	FACILITY NAME	DOCKET NUMBER
										05000
OPERATING MODE (9)		1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)		071	20.2201(b)			20.2203(a)(2)(v)			50.73(a)(2)(i)	
			20.2203(a)(1)			20.2203(a)(3)(i)			50.73(a)(2)(ii)	
			20.2203(a)(2)(i)			20.2203(a)(3)(ii)			50.73(a)(2)(iii)	
			20.2203(a)(2)(ii)			20.2203(a)(4)		X	50.73(a)(2)(iv)	
			20.2203(a)(2)(iii)			50.36(c)(1)			50.73(a)(2)(v)	
			20.2203(a)(2)(iv)			50.36(c)(2)			50.73(a)(2)(vii)	
									OTHER	
									Specify in Abstract below or in NRC Form 366A	

LICENSEE CONTACT FOR THIS LER (12)

NAME

John T. St. Martin - Technical Assistant

TELEPHONE NUMBER (Include Area Code)

(716) 771-3641

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	KE	EXC	S245	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE).		X	NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 21, 2000, at approximately 1418 EDST, with the plant in Mode 1 at approximately 71% steady state reactor power, the "B" condenser circulating water pump tripped. At approximately 1419 EDST, following procedural direction, the reactor was manually tripped. The Control Room operators performed the appropriate actions of procedures E-0 and ES-0.1. Following the reactor trip, all systems operated as designed, and the reactor was stabilized in Mode 3.

The tripping of the "B" condenser circulating water pump was caused by failure of the excitation voltage transformer, which resulted in loss of excitation field to the pump motor. The cause of the reactor trip was manual operator action.

Corrective actions included replacing the excitation voltage transformer.

Corrective action to prevent recurrence is outlined in Section V.B.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)**I. PRE-EVENT PLANT CONDITIONS:**

On October 21, 2000, at approximately 1418 EDST, the plant was in Mode 1, holding at approximately 71% steady state reactor power, during the initial power ascension after completion of the 2000 refueling outage. Reactor coolant system (RCS) temperature was being maintained at approximately 555 degrees F and pressurizer pressure at approximately 2235 psig. Control rods were in "manual" and power ascension was limited to 3% per hour for preconditioning of new fuel loaded during the 2000 refueling outage. Plant electrical output was limited due to pre-planned offsite power line maintenance. Reactor protection system (RPS) axial offset calibration was in progress for RPS channel I.

II. DESCRIPTION OF EVENT:**A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:**

- October 21, 2000, 1418 EDST: "B" condenser circulating water (CW) pump trips.
- October 21, 2000, 1419 EDST: Control Room operators manually trip the reactor, verify both reactor trip breakers open, and verify all control and shutdown rods inserted. Event date and time.
- October 21, 2000, 1419 EDST: Discovery date and time.
- October 21, 2000, 1420 EDST: Both main feedwater pumps automatically trip on low seal water differential pressure.
- October 21, 2000, 1426 EDST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- October 21, 2000, 1441 EDST: Plant is stabilized in Mode 3.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)**B. EVENT:**

On October 21, 2000, at 1418 EDST, the plant was in Mode 1 at approximately 71% steady state reactor power. The Control Room operators received Main Control Board Annunciator J-16 (Motor Off CW-EH Emerg Oil Seal Oil BU), caused by the trip of the "B" condenser circulating water (CW) pump. The trip of the "B" CW pump was followed by closure of the discharge valve for the "B" CW pump. The Control Room operators observed that the "B" CW pump had tripped, entered Abnormal Operating Procedure AP-CW.1 (Loss of a Circ Water Pump), and performed the appropriate actions.

Procedure AP-CW.1 provides direction for loss of a CW pump. In accordance with AP-CW.1, a manual reactor trip was ordered by the Control Room Foreman. The Control Room operators manually tripped the reactor at approximately 1419 EDST and performed the appropriate actions of Emergency Operating Procedure E-0 (Reactor Trip or Safety Injection). They transitioned to Emergency Operating Procedure ES-0.1 (Reactor Trip Response) when it was verified that both reactor trip breakers were open, all control and shutdown rods were inserted, and safety injection was not actuated or required.

During the subsequent loss of load for the secondary system, main feedwater (MFW) pump suction pressure increased above its alarm setpoint. Main Control Board (MCB) annunciator H-11 (Feed Pump Seal Water Lo Diff Press 15 PSI) also alarmed, indicating that the high suction pressure had eliminated the differential pressure across the MFW pump seals. Both MFW pumps automatically tripped on low seal water differential pressure at approximately 1420 EDST.

During the transient, level in the "A" steam generator (SG) decreased below 17%, which resulted in automatic start of both motor-driven auxiliary feedwater (AFW) pumps. The Control Room operators verified that the AFW pumps had started as designed on Lo Lo SG level. The Control Room operators received Main Control Board Annunciator K-3 (AMSAC Actuation) at approximately 1421 EDST (due to 3/4 feedwater flow channels <25%) and verified that the turbine-driven AFW pump had started due to a signal from the ATWS Mitigation System Actuation Circuitry (AMSAC). During the performance of ES-0.1, the Control Room operators noted that a reactor coolant system (RCS) cooldown was occurring and manually stopped the turbine-driven AFW pump at approximately 1422 EDST. They manually closed both main steam isolation valves (MSIV) at approximately 1426 EDST. These procedurally-directed actions mitigated the RCS cooldown.

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The plant was stabilized in Mode 3 at approximately 1441 EDST, and the Control Room operators transitioned to normal operating procedures.

The "B" CW pump was restored to service at approximately 2242 EDST on October 21, 2000, after replacement of the failed excitation constant voltage transformer and inspection of the associated electrical circuitry for the motor.

A reactor trip (referred to as "scram" in NRC Performance Indicators) occurred. This scram meets the definition for the NRC Performance Indicator (PI) "Unplanned Scrams Per 7,000 Critical Hours". The scram does not meet the definition for the NRC PI "Scrams With a Loss of Normal Heat Removal" since the normal heat removal paths (as listed in NEI 99-02, Revision 2) were removed due to intentional operator actions to control the reactor cooldown rate.

C. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

E. METHOD OF DISCOVERY:

The trip of the "B" CW pump was immediately apparent to the Control Room operators due to Main Control Board (MCB) Annunciator J-16 and MCB indicating lights for the "B" CW pump. The reactor trip was manually initiated and was confirmed by plant response, alarms, and indications in the Control Room.

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F. OPERATOR ACTION:

The Control Room operators promptly identified the loss of the "B" CW pump and performed the appropriate actions of AP-CW.1. They initiated a manual reactor trip as directed by procedures. After the reactor trip, the Control Room operators performed the appropriate actions of procedures E-0 and ES-0.1. They promptly secured the TDAFW pump within a minute of its starting, and manually closed the MSIVs approximately six (6) minutes after the trip to limit further RCS cooldown. The plant was stabilized in Mode 3.

The Control Room operators subsequently notified higher supervision. The shift supervisor notified the NRC per 10CFR50.72 (b) (2) (ii), non-emergency four hour notification, at approximately 1758 EDST on October 21, 2000.

G. SAFETY SYSTEM RESPONSES:

All safeguards equipment functioned properly. Both motor-driven AFW pumps started when SG level decreased below 17% after the reactor trip. The turbine-driven AFW pump started as per design, due to a starting signal from AMSAC. This condition does not meet the definition for the NRC Performance Indicator (PI) "Safety System Functional Failure" because all safety systems functioned as designed.

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the reactor trip was manual trip initiation, ordered by the Control Room Foreman as directed by procedure AP-CW.1 in response to loss of the "B" CW pump.

B. INTERMEDIATE CAUSE:

The intermediate cause of the loss of the "B" CW pump was low power factor.

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C. ROOT CAUSE:

The underlying cause of the trip of the "B" CW pump was the tripping of the power factor protective relay for the "B" CW pump motor. This relay was activated due to a decrease in the power factor for this motor, caused by failure of the "B" CW pump excitation SOLA constant voltage transformer (CVT), which resulted in a loss of excitation field to the "B" CW pump's synchronous motor. The CVT failed due to overheating. The practice of maintaining the excitation CVT energized when the associated CW pump is not running contributed to long-term degradation of the CVT. This event is NUREG-1022 Cause Code (B), "Design, Manufacturing, Construction / Installation".

IV. ANALYSIS OF EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (iv), which requires a report of, "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)". The manual reactor trip is an actuation of the RPS, and AFW pump starts are actuations of an ESF.

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

There were no operational or safety consequences or implications attributed to the trip of the "B" CW pump and subsequent manual reactor trip because:

- The two reactor trip breakers opened as required.
- All control and shutdown rods inserted as designed.
- The plant was stabilized in Mode 3.

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- The Ginna Updated Final Safety Analysis Report (UFSAR) was reviewed. This transient is bounded by a total loss of external electrical load accident. Results of the analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures within the design limits. The integrity of the core is maintained by operation of the reactor trip system (RTS); i.e., the DNBR is maintained above the limit value.

This UFSAR transient was examined and compared to the plant response for the actual event. The plant behavior was found to be consistent with, and bounded by, the event detailed in the accident analysis.
- The Ginna Improved Technical Specifications (ITS) Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) were reviewed with respect to the post trip review data. The following are the results of that review:
 - Pressurizer (PRZR) pressure decreased below 2205 PSIG during the transient after the reactor trip. During this time the plant was in Mode 3, where ITS LCO 3.4.1 is no longer applicable. Therefore, compliance with ITS was maintained. Additional mitigation was provided by stopping the turbine-driven AFW pump and closing the MSIVs. Minimum PRZR pressure was approximately 2125 PSIG, and PRZR pressure was restored > 2205 PSIG within approximately four (4) minutes.
 - After the reactor trip, the RCS cooled down to approximately 540 degrees F and was subsequently stabilized at 547 degrees F. The cooldown was within the limits of ITS LCO 3.4.3. In addition, the required shutdown margin was maintained at all times during the RCS cooldown.

Based on the above and the review of post trip data and past plant transients, it can be concluded that the plant operated as designed, that there were no unreviewed safety questions, and that the public's health and safety was assured at all times.

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V. CORRECTIVE ACTION:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

- The Control Room operators performed the appropriate actions of Emergency Operating Procedures E-0 and ES-0.1 and the plant was stabilized in Mode 3.
- The SOLA CVT for the "B" CW pump was replaced.
- The power factor protective relay for the "B" CW pump motor was inspected and tested satisfactorily.
- The "B" CW exciter was tested and found satisfactory. CW motor test data was reviewed and found satisfactory.
- The CVT for the "A" CW pump was inspected.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

NOTE: There are no NRC regulatory commitments in this Licensee Event Report.

- Ventilation air flow has been enhanced in the compartment where the SOLA CVT is located.
- To avoid long-term degradation of the CVT, procedures that address operation of the CW pumps will be revised to require that the circuit breaker that supplies power to the excitation CVT be opened when the associated CW pump is not running.

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VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The "B" CW pump motor is a Westinghouse "Life Line" series motor, Model # 110P662H01, frame size HR-111-SPL, rated for 4000 volts and 1760 horsepower.

The CVT is harmonic neutralized type constant voltage sinusoidal CVT, catalog number 23-23-210-8, rated for 1000 VA (volt-amps), and is manufactured by SOLA.

B. PREVIOUS LERs ON SIMILAR EVENTS:

An historical search of LERs was conducted with the following results: No documentation of similar LER events with the same root cause could be identified. However,

- LER 95-008 was a similar event (loss of CW pump, resulting in a manual reactor trip) with a different root cause for the loss of the CW pump. The corrective action to prevent recurrence would not have prevented LER 2000-005.
- LER 96-002 was a similar event (loss of CW pump, resulting in a manual reactor trip) with a different root cause for the loss of the CW pump. The corrective action to prevent recurrence would not have prevented LER 2000-005.
- Recent reactor trips with different root causes are LER 96-012, LER 1999-007, LER 1999-008, and LER 2000-001

C. SPECIAL COMMENTS:

None

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D. IDENTIFICATION OF COMPONENTS REFERRED TO IN THIS LER:

COMPONENT	IEEE 803 FUNCTION	IEEE 805 SYSTEM IDENTIFICATION
circulating water pump	P	KE
auxiliary feedwater pump	P	BA
main feedwater pump	P	SJ
main steam isolation valve	ISV	SB
steam generator	SG	SB